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Abstract

Evaluation of Post-Incident Cooling Systems of Light Water Power Reactors

Hideo Uchida, Akira Oyama, Yasumasa Togo
(University of Tokyo)

Significance of post-incident cooling systems is evaluated in connection with so-called "maximum credible accident (MCA)" of light water power reactors.

In evaluation of a core spray system, non-uniformity of core spray flow distribution is taken into account. Introducing experimental data of core spray flow distribution and spray heat transfer, quantiative calculations are made of percentages of fuel segments ruptured in case of loss-of-coolant accidents. Calculations are made on JPDR (BWR, 12.5 Mwe) design data.

Stability criteria for spray cooling are also discussed. In evaluation of a container spray system, reduction factors of F.P. release are calculated, and relative significance of container spray systems is discussed.

In order to experimentally evaluate post-incident cooling systems (core spray, safety injection and container spray systems), Project SAFE (Safety Assessment and Facilities Establishment Project) has been established in Specialists Committee for Study on Nuclear Safety of Japan Atomic Industrial Forum. Specialists in various fields (universities, research laboratories and manufacturers) attend this Committee. During first period of this Project (from April 1963 to September 1964), amounts of flushed coolant at loss-of-coolant accidents are evaluated by a non-nuclear model, as well as temperature and pressure transients of a container. It is also aimed to run basic experiments on effects of types of spray nozzles and its arrangement, spray heat transfer and wash-out effect, for the purpose of comparing calculation results with experimental data. Experimental results obtained so far are reported.

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Abstract

The Capture of the Radioactive Noble Gases by the Clathrate Method

By

Hikaru Shimojima, Y. Nakayama, K. Matsumoto and H.Hyodo

NAIG Nuclear Research Laboratory

250, Suehiro-cho, Kawasaki, Japan

It is important to treat safely the radioactive noble gases released from atomic power plants and the fuel reprocessings. These gases have been conventionally treated by the charcoal bed method, but, since these gases can't keep adsorbed with stability under the ordinary temperature and pressure, the method is not convenient for the storage and transport of the gas products.

Considering these shortcomings of the charcoal bed method pointed out above, the authors studied for the development of a new unit operation technique of the waste disposal of the radioactive noble gases using clathrate compounds. Laboratory scale experiments were performed in order to determine conditions, under which industrial scale processes could be made.

Hydroquinones are combined with each other by the intermolecule hydrogen bonds to make cages which have the suitable size for including other molecules such as noble gases and then make compounds of $3C_6H_4(OH)_2 \cdot M$ type which are called clathrate compound. From the formula represented above, the clathrate compounds prove to contain the noble gases 97.4ml.S.T.P./g in maximum, so that they have the absorption capacity compatible with that of the charcoal bed. They are not only convenient for storage and transport but appropriate to wide industrial applications, since they are stable under the ordinary temperature and pressure and decomposed on heating to release the radioactive noble gases.

In this work, various amounts of solutions saturated with hydroquinone at various temperatures were heated up to several degrees above the saturation tempe-

rature in atmosphere of the noble gases under various pressures, and then cooled to the room temperature at constant cooling rates. While the crystal was growing, the noble gases entered the cages of the clathrate compounds as mentioned above.

Of the nine solvents experimented, which were water, and eight other organic solvents which dissolve noble gases more readily than water, the first one gave the best result.

The experiments were performed in aqueous solution in detail to study important factors which have influences on crystal growth, (i) super solubility with and without crystal seeds and also in existance of other substances such as CH_3OH, CO_2 which can also be included in the clathrate compounds, (ii) stability of super cooled solutions, and (iii) the rate of crystal growth.

Two types of reactors, a batch type autoclave and a semi-continuous gas flow type reactor were used to investigate the effect of the experimental conditions on the noble gas contents in the clathrate compounds and the decontamination factor. The following experimental conditions were considered; the partial pressure of the noble gases, the flow rate of the noble gases, the cooling rate of the solution and the saturation temperature of the hydroquinone solution.

The results obtained from the experiments imply the possibility of the safe disposal of the radioactive noble gases by the clathrate method, since the gases can be compressed and held in a small space in place of dilution or mixing with other gases.

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Abstract

AN EXPERIMENTAL STUDY ON THE MECHANICAL STRENGTH OF FUEL CLADDING TUBES UNDER THE LOSS-OF-COOLANT ACCIDENT CONDITIONS OF WATER COOLED REACTORS

I. Katsura, Tokyo Shibaura Electric Co., Ltd.
M. Kosaki, Sumitomo Metal Industries, Ltd.
Y. Mishima, The University of Tokyo
K. Ono, Hitachi, Ltd.
Z. Takao, Kobe Steel Works, Ltd.
H. Tanaka, The Furukawa Electric Co., Ltd.
M. Toyoda, The Japan Atomic Power Co.
S. Wakisaka, Tokyo Electric Power Co., Inc.
M. Yokosuka, Mitsubishi Atomic Power Industries, Inc.
of the Specialists Committee for Study of Nuclear Fuel Safety of Japan Atomic Industrial Forum, Inc. (Chairman: Y. Mishima)

The purpose of this experiment, which has been carried out by a group of specialists under auspices of Japan Atomic Industrial Forum, is to review and confirm the mechanical integrity and behavior of fuel element cladding tubes for light water reactors under the loss-of-coolant accident conditions.

Accumulation, to a certain extent, of fission gases within fuel elements will not affect cladding tubes as the source of internal pressure while the reactor is under operation, since the coolant outside the tubes offsets it as an external pressure. If the coolant pressure drops by such a reason as the loss-of-coolant accident, however, an effective difference between pressures in and out the tubes rises instantaneously, and this may result in bursting. Temperature rise due to the decay heat after the loss-of-coolant causes internal pressure to rise, which may also result in bursting. Evaluation of fuel cladding safety under simulation of such conditions was made by this experiment.

The AISI 304L, 347 stainless steel and Zircaloy-2 tubes were selected for the testing and their diameter-to-thickness ratios were selected between 12.0 and 28.7. Hydrogen absorbed Zircaloy-2 tubes specimens

were also selected for evaluating the effect of hydrogen absorption on the mechanical behavior under the accident conditions.

The experimental conditions were determined from the analysis of the loss-of-coolant accidents for the typical pressurized as well as boiling water reactors. Experiments were performed under the following conditions:

1. Tensile test of the cladding materials at high temperatures up to 700°C, and at the various moderate load rates up to 5.0 $kg/mm^2/sec$.

2. Burst test of the various sized cladding tubes for temperatures ranging from the room temperature up to 700°C.

3. In the dynamic burst test under the various loss-of-coolant accident conditions, the specimens were internally pressurized to the predetermined value at the surface temperatures between 280°C and 450°C and under the external pressure corresponding to the coolant pressure, and then the external pressure was suddenly reduced at the postulated rate for the loss-of-coolant accident.

It is concluded from a series of the above experiments that the mechanical properties of the cladding tubes are scarcely affected by such a load change of postulated pressure decreasing rate at the loss-of-coolant accident, and also it is experimentally confirmed to be possible to estimate the bursting pressure of the fuel cladding tube from the tensile strength of the materials. Less expansion was observed on the Zircaloy-2 tube after bursting which absorbed about 300 ppm hydrogen, though its tensile strength was found to have little changed.

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Abstract

SOME JAPANESE WORKS ON CHEMICAL SEPARATION OF URANIUM ISOTOPES

H. Kakihana, Tokyo Institute of Technology, Tokyo, Japan
Atomic Fuel Corporation, Tokyo, Japan
T. Hoshino, H. Sato and K. Gonda, Atomic Fuel Corporation, Tokyo, Japan
Y. Mori, K. Kurisu and T. Kanzaki
Tokyo Institute of Technology, Tokyo, Japan
J. Shimokawa and G. Nishio
Japan Atomic Energy Research Institute, Tokyo, Japan
H. Goto and S. Suzuki, University of Tohoku

Part I. A Theory of Uranium Isotope Separation involving Electron Exchange Reaction

The theoretical studies have proved that the mixture of U(IV) and U(VI) should be a favorable system, if the electron exchange reaction between the isotopes could be catalysed.

Part II. Ion Exchange Resin as an Effective Catalyzer to the Electron Exchange Reaction Between U(IV) and U(VI)

Cation and anion exchange resins were ascertained as very promising catalyzers. For example:

$$\frac{\text{rate of electron exchange in cation exchange resin}}{\text{rate of electron exchange in solution}} = 10^3 - 10^4$$

Part III. Chemical Separation of Uranium Isotope with Ion Exchanger in the System containing both U(IV) and U(VI)

Better separation factors were observed for U(IV) (1.0007) than U(VI) (1.0000). Much higher separation factors (1.0014) were obtained for the mixed systems of U(IV) and U(VI), which could be explained by the effective contribution of the isotopic effect in electron exchange reaction between U(IV) and U(VI).

Part IV. Uranium Isotope Fractionation in Ion Exchange Membrane by Ionic Migration

At the front of U-band in ion exchange membrane an appreciable enrichment of 235-U was observed just as in the case of Li isotcpe.

Part V. Effect of Superposing A.C. on Isotope Separation in Ion Exchange Resin Column by Ionic Migration

By superposing A.C. similar favorable effects to N isotopes were observed on U isotope fractionation.

Part VI. A New Continuous Isotopes Separation Apparatus Based on a Simultaneous Use of Ion Exchange Elution and Ionic Migration

An apparatus based on the idea of combining two processes, ion exchange elution and ionic migration, was devised to aim the synergetic effect of them as well as the continuous operation. Through this apparatus the continuous flow of enriched 6-Li (e.g. 7-Li/6-Li = 11.61) was obtained at the cathode side, while the depleted 6-Li (e.g. 12.81) at the anode side. Similarly the enrichment of 235-U was observed in the flow of the cathode side.

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Abstract

ISOTOPE SEPARATION BY CONCURRENT GAS CENTRIFUGE

A. Kanagawa, Y. Takashima, and Y. O yama*

* Research Laboratory of Nuclear Reactor, Tokyo Institute of Technology

The relations between the separative power, the separation factor, the gas flow rate, the length of cylindrical rotor, the cuts and the radii to be extracted have generally been analized concerning isotope separation by the concurrent type of gas centrifuge and compared with those in the case of the countercurrent type.

The more generalized differential equation than that used by Cohen, which can be applied to centrifugal separation through (n+1) cylindrical thin concurrent streams, has been obtained, and solved in the cases of n=1 and n=2.

First of all, in the case of n=1, which already solved by Cohen, the relations of gas feed rate per length of the rotor to both the separation factor and the separative power have been calculated and compared with those in the case of countercurrent flow using a parameter of Z/r_0 which is the ratio of the length of rotor to the radius. As a result, it has been cleared quantatively that, as Z/r_0 increases, remarkably more flow rate should be required in the concurrent flow to get the same value of separative power than that in the countercurrent flow, and the flow rate in the concurrent flow, however, is comparable with the circulation rate in the countercurrent flow.

In the case of n=2, the higher maximum separative power is possible to be realized even if the inner two thin streams are mixed at the exit, than that in the case of n=1. The optimum condition in this case is as follows; $r_1/r_0=0.45$, $r_2/r_0=0.6$, $\theta_1=0.19$ and $\theta_2=0.31$, where r_1 and r_2 are the radii of inner two streams and θ the cut. The maximum value of the separative power is 1.066 times of that in the case of n=1, then the optimum flow rate is 1.130 times.

As a result of these analyses, it will be suggested that the thickness of the flow path which should exist more or less in practice may bring rather preferable separation performance.

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Abstract

IN-PILE EXPERIMENTAL EQUIPMENT FOR BR2

J. Planquart
Head, Technology Department, C.E.N.

H. Knaab
Engineer, "Siemens" (Erlangen)

A. Lhost
Engineer, "BelgoNucléaire" (Brussels)

W.R. Ruston
Manager, S.E.R.A.I. (Brussels)

M. Soenen
Engineer, Technology Department, C.E.N.

G. Vanmassenhove
Engineer, Technology Department, C.E.N.

Centre d'Etude de l'Energie Nucléaire (C.E.N.)
MOL (Belgium)

and

Société d'Etudes, de Recherches et d'Applications pour l'Industrie (S.E.R.A.I.)
Brussels (Belgium)

A large part of this work has been performed by the Association C.E.N.-EURATOM for the exploitation of the BR2 reactor and its auxiliary installations (Contract N°006-60-5/BRAB).

The paper gives a description of some of the principal in-pile irradiation devices for use in the high flux research reactor BR2; points of discussion deal mainly with the application of engineering principles to the problems encountered in submitting both fissile and non-fissile materials to a neutron flux.

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Several loops and rigs with or without instrumentation are reviewed.

A sodium cooled loop for the test irradiation of specimen fast reactor fuel pins in a fast neutron flux has been designed for the French Commissariat à l'Energie Atomique. The loop is intended for the study of the behaviour of the fuel pins under conditions as near as possible to those in a sodium cooled fast reactor. The characteristics which have been reproduced, are the mass flow rate of sodium, the neutron spectrum and the specific power output (2 kW/cm^3). On the other hand, much thought had to be given to the problems of the safety aspects, the operation and the availability to other users of the BR2 reactor while the loop is in-pile.

A paricular point of interest of the loop is that the primary sodium circuit is very compact ; it is contained in a pressure tube which constitutes the in-pile section and therefore does not extend outside the BR2 reactor pressure vessel.

To obtain the required neutron spectrum, two annular neutron filters, one of cadmium and one of boron carbide, placed in series "filter out" the low energy neutrons. To achieve the required neutron flux the fuel elements of the BR2 core will entirely surround the in-pile section of the loop.

Design and manufacturing problems connected with the loading of the loop had to be solved; these problems arise not only from the loop itself but from the difficulties associated with the accomodation of the loop in the reactor. The paper examines the range of the problems met to operate the loop in-pile.

A gas cooled loop designed for the O.E.C.D. Dragon project has been put into operation to irradiate in a high flux reactor samples of graphite of a type to be used in a high temperature gas cooled reactor.

The aim of the research programme is to irradiate the graphite in a high neutron flux in an extended temperature range. A very important section of the ancillary equipment of the loop is the helium purification system because a great amount of research interest lies in the phenomenon of mass transfer of the graphite to the helium and the influence on this of the impurities conveyed by the helium.

The loop includes several novel features notably those associated with the design and manufacture of the in-pile section.

The paper examines the problems of building and operating the loop.

A gas cooled loop, principally for testing fuel elements for gas cooled reactors is designed for SIEMENS (Erlangen) by S.E.R.A.I. One may check, for instance, the dimensional stability of the fuel and the variation of thermal conductivity under irradiation, study the diffusion of fission products and the influence of thermal cycling, find out the life-time (maximum burn-up of the elements), as well as the heat flux corresponding to the partial or total melting of the fuel, to the growth of the grains and to the failure of the cladding. The experiments may also yield interesting data on the corrosion behaviour of the samples as a function of coolant composition.

Besides testing fuel elements, the flexibility of the loop is such as to allow other experiments amongst which the examination of different structural or control materials under irradiation, as well as investigation on coolant chemistry. The study of the graphite-CO_2 reaction at high temperature and under irradiation seems to be particularly important.

The loop may also help to check efficient detection systems to indicate contamination of reactor circuits. Decontamination methods and general equipment behaviour can also be investigated.

The general loop lay-out and the design of its different sections are discussed in detail.

Other experimental equipment is either under consideration or developed for the BR2 reactor. Each experiment presents some particular problems which have to be solved before it can go in-pile. Initially each experiment is examined from the nuclear and thermal characteristics standpoints before a complete design study is started. Eventually nuclear and thermal calculations are verified by measurements on models in the BRO2 and the BR2, for fine and coarse reactivity calibration, and by doing heat balances out-of-pile.

Among the rigs designed attention is drawn to a boiling water capsule in which knowledge of the heat flux from a fuel pin may be obtained.

The paper gives a review of some particular applications of rigs and also describes improvements and modifications carried out on the BR2 fuel elements to provide better irradiation facilities.

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Abstract

DESIGN AND FABRICATION OF PWR FUEL ELEMENTS IN BELGIUM

M. HUBERLANT, Ingénieur Chef de Département, and J. WERTZ, Ingénieur Chef de Service à la S.A. Métallurgie et Mécanique Nucléaires (M.M.N.), Dessel.

A. de STREEL and J. VAN MULDERS, Ingénieurs Chefs de Service à la S.A. Bureau d'Etudes Nucléaires (B.E.N.), Brussels.

Belgium is particularly interested in the development of uranium oxyde PWR reactors.

This trend has been followed since the decision to build the BR-3 and SENA reactors. The development of the VULCAIN project has stimulated the interest in this type of reactor.

In this field, M.M.N. has developed and fabricated up to now :

- the fuel loading for the VENUS critical experiment in relation with the VULCAIN project,
- the loading of the VULCAIN power exepriment in the BR-3 and
- a prototype fuel element for the BR-3 reactor.

The latter element, built according to the latest design and engineering methods evolved for the SENA core, includes twelve plutonium fuel rods developed by the Association BelgoNucleaire-C.E.N. and is of a different type than the original elements. It has been introduced in the central part of the BR-3 core amongst elements of the original design.

B.E.N. has carried out the theoritical studies relative to its behaviour in the reactor from the various nuclear, hydraulic and thermal points of view.

The problems encountered in the fabrication and the study of this fuel

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element are the subject of this paper.

The fabrication is based on the following principles : the fuel rods, consisting of uranium oxyde pellets enclosed in thin walled stainless steel tubes, form a bundle held in an independent perforated can, allowing free thermal expansion of the rods. The tubes are positioned by six intermediate spacing grids with integral spring clips. Stainless steel is used for the tubes, can and grids.

The element will be irradiated until depletion of the present BR-3 core, corresponding to a burnup of approximately 5.000 MWD/T for the experimental fuel element. This test of about 4,000 full power hours will provide a check of the mechanical behaviour of this type of element in a power reactor.

The introduction of an element so different from the original ones results in an heterogeneity which in general tends to reduce the maximum power capability of the core. It was therefore necessary to perform a theoretical study to verify the operating safety of the new core configuration at the plant rated power. The following aspects have been investigated :

- The nuclear study has for objective the determination of the local power peaks and their variations during core life.
- The hydrodynamic study analyzes the coolant flow distribution between the various coolant channels.
- The thermal study combines the results of the two previous studies to locate the thermal hot channels. Its main purpose is to check, by means of the latest DNB (Departure from Nucleate Boiling) correlations published by Westinghouse, the operating safety of the core at rated power.

The operation of the reactor at full power, without any incident due to the presence of the new fuel element, has corroborated the suitability of the fabrication methods and the validity of the theoretical results.

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Abstract

PLUTONIUM ENRICHED CERAMIC FUEL RODS IRRADIATION IN THE BR 3 POWER REACTOR

J. M. Leblanc
Chef de service adjoint à la BelgoNucléaire

Plutonium Laboratory
Société Belge pour l'Industrie Nucléaire S. A. (BelgoNucléaire)
Centre d'Etude de l'Energie Nucléaire (C. E. N.)
MOL (Belgique)

Work performed in the frame of the Joint U.S./EURATOM Research Program

Twelve plutonium - uranium oxide fuel rods have been introduced in the Belgian BR 3 reactor. The main objective of this experiment was to raise and to solve the technological problems concerning the fabrication of plutonium enriched fuel rods and to study their behaviour under the pressurized water power reactor operation conditions.

The fuel rods have been prepared by vibratory and swaging compaction of physically mixed dense UO_2 and UO_2 - PuO_2 powders. The main characteristics of these fuel rods are the following :

- Fuel material : UO_2 - 0,96 m/o PuO_2
- Density : vibrated rods : 86 % T D
  swaged rods : 91 % T D
- Dimensions : length : 150 cm

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outside diameter : 0,870 cm
thickness of the cladding : 0,050 cm
(stainless steel AISI 348)

At each fabrication step, destructive and non destructive controls have been made to estimate the quality of the used materials. The non destructive tests realized on the finished fuel rods essentially consist in controlling the end cap weldings and the canning thightness, the homogeneity of both the density and the plutonium enrichment, the integrity and the thickness of the cladding, the surface contamination, etc...

The twelve plutonium enriched rods have been distributed among 96 rods containing uranium oxide enriched to 3,7 % U^{235} and assembled with these latter by means of grids provided with springs (1). The assembly has been introduced in the core center in the beginning of November 1963. It will remain in the reactor core during at least 9 months (4000 hours at the equivalent full power of 40,9 MW th).

The computation of the plutonium enrichment (0,96 m/o) in the rods have been done by use of a geometrical model idealizing the reactor core when the fuel rods are introduced. The safety criterium adopted is the uniformity of the radial specific power distribution in the core. The nuclear study made it possible to estimate the maximum specific power generated by each rod at the beginning and at the end of the core life. At the standard power of 40,9 MW th, the maximum specific power in the plutonium loaded rods is about 330 W/cm, the average burn-up of the most highly rated rod will be close to 4800 MWd/t.

Prior to the introduction of the fuel rods in the reactor, different

researches have been carried out. Out of pile experiments have been made in order to determine the thermal conductivity of physically mixed UO_2 - PuO_2 powders densified by vibration and swaging. From the results of the measurements, the maximum central temperature of 1900° C $\pm$75° C has been evaluated in the plutonium fuel rods during the BR 3 operation. Fuel samples prepared by means of the same techniques and irradiated in the BR 2 high flux reactor have allowed to conclude that such a type of fuel may tolerate thermal heat fluxes of about 380 - 400 W/cm and burn-ups close to 1000/1100 MWd/t.

The reactor is actually in operation since November 25, 1963. The experiments still going on are one of the first objectives pursued by the BelgoNucléaire - C. E. N. Association researches on plutonium.

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(1) The U^{235} enriched fuel rods as well as the subassembly have been realized by the Belgian firm "Métallurgie et Mécanique Nucléaires" (MMN).

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Abstract

PREPARATION AND PHYSICAL PROPERTIES OF UO_2 SINGLE CRYSTALS

P. Nagels W. Van Lierde R. Strumane R. Penninckx L. Mewissen
J. Devreese R. De Coninck R. De Batist J. Bressers
M. Denayer L. De Jonghe M. De Coster
H. Blank
Research and Information Division, Euratom, Brussels
S. Amelinckx
Head, Solid State Physics Department
Centre d'Etude de l'Energie Nucléaire (C.E.N.)
MOL (Belgium)

Work done under contracts C.E.N.-EURATOM N°063-61-10/RDB and 010-63-3/TEEB

An extensive study of different fundamental physical properties of UO_2 single crystals of different compositions has been undertaken. The different properties were studied before and after reactor irradiation. This work includes the preparation of single crystals, the variation of composition of the single crystals by controlled oxidation, sintering of single crystal spheres, transmission electron microscope observation of thin foils and also the measurement of the thermal conductivity, the electrical conductivity, the thermoelectric power, the optical absorption spectrum, the elastic constants and of the resonant γ-ray absorption.

The results obtained as yet are reviewed in this communication. A method for growing uranium dioxide single crystals by sublimation has been developed. High quality semi-spherical crystals from 4 to 15 mm diameter, up to 7 g in weight, can be prepared. The stoichiometric oxygen excess of the UO_{2+x} crystals can afterwards be varied in the range $0 < x < 0.25$ by carefully controlled oxidation. This was achieved by a new, simple and reliable method.

UO_2 single crystals can be prepared as thin foils suitable for transmission electron microscopic observation. The thinning is achieved chemically at 200°C or electrolytically at room temperature. Dislocations, precipitates of different shapes, consisting presumably of U_4O_9, fission fragment damage, as well as tracks, clusters of point defects and small dislocation loops, are observed and discussed.

It has been possible to cut square elongated prisms of about 8 mm in length from the vapour grown single crystals. These prisms are suitable for the determination of the thermal conductivity. The latter has been measured from liquid helium temperature up to room temperature. No striking qualitative deviation has been found from the results obtained by other investigators using sintered polycrystalline material.

The electrical conductivity and the thermoelectric power of UO_2 single crystals of different composition have been measured in an extended temperature range before and after reactor irradiation. The latter results in a very marked decrease of the conductivity of the non-stoichiometric crystals. This effect can be attributed to the formation of hole traps during irradiation. From recovery experiments, one concludes to the existence of an annealing peak around 725°C, the associated activation energy being about 2.5 eV. This suggests possible uranium point defect migration.

From the analysis of the optical absorption spectrum, a value of 2.18 eV is found for the absorption edge at room temperature.

The elastic constant has been determined using a pulse echo technique, which allows a precision of about 1% .

UO_2 doped with Fe^{57} shows an unsplit single peak Mössbauer spectrum. However, if kept for nine months at room temperature a spectrum appears which can be analysed as arising from the superposition of this peak and those arising from magnetic hyperfine splitting.

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Abstract

CORROSION OF REACTOR MATERIALS

Ph. Berge
Project Supervisor, Euratom, Brussels

M. J. Brabers
Assistant Head, Metallurgy Department, C.E.N.

J. Hennaut
Searcher, Physical Methods Division, S.E.R.A.I.

M. Maurice
Assistant, Physical Methods Division, S.E.R.A.I.

G. Mestdagh
Engineer, Metallurgy Department, C.E.N.

J. Meunier
Delegate of the "Centre Belge d'Etude de la Corrosion"(CEBELCOR)
Brussels

C. Sonnen
Assistant, Physical Methods Division, S.E.R.A.I.

G. Spaepen
Engineer, Metallurgy Department, C.E.N.

M. Warzée
Head, Physical Methods Division, S.E.R.A.I.

J. Waty
Searcher, Physical Methods Division, S.E.R.A.I.

P. Weisgerber
Fellow from Euratom "Centre Commun de Recherches", Ispra, Italy

Société d'Etudes, de Recherches et d'Applications pour l'Industrie, (S.E.R.A.I.)
Brussels (Belgium)

and

Centre d'Etude de l'Energie Nucléaire (C.E.N.)
MOL (Belgium)

Work performed under contracts S.E.R.A.I.-EURATOM/U.S.A. and C.E.N.-EURATOM/U.S.A. N°089/62/7/RDB and 084/62/3/RDB.

The results of the work are presented in two parts.

1. An electrochemical study of the effect of irradiation on the corrosion of some reactor materials.

Corrosion experiments on aluminium, zirconium, titanium, niobium, zircaloy-2, stainless steel and platinum have been carried out in-pile, and after pile irradiation, and the results have been compared with those obtained out-of-pile under identical conditions. Electrochemical techniques were applied as testing method. The in-pile measurements consisted either in measurement of the time necessary to build up a potential at a given constant electric current (transitometric method) or in measurement of the electric current as a function of changing potential (potentiodynamic method). The post-irradiation experiments were done by oscillographic recording of the overpotential versus the time when a single charge pulse was supplied to the specimen. The in-pile measurements were done in aqueous solutions, the irradiation of specimens for the post-irradiation examination in mixtures of dried argon and oxygen and in high purity argon.

In the case of aluminium it was found that the passivation properties of the oxide layer formed during pile irradiation in aqueous solutions were superior to that formed in the absence of radiation.

Potentiodynamic measurements on steel 316L and on platinum showed that in acid, neutral and alkaline solutions the zero-current potential during irradiation shifts to the cathodic direction. When the solution only is irradiated a similar shift occurs. This may indicate that by irradiation the solutions become relatively more reducing. The anodic current with increasing potential is under irradiation in all solutions higher than in the absence of irradiation. This increase was most pronounced in acid solutions and least in alkaline solution. The nature and quantity of dissolved gases have some importance but the effects are not well understood.

The effect of chloride concentration on pitting corrosion of steel has been studied. The results indicate that in-pile the onset of pitting attack is retarded in comparison with out-of-pile experiments.

The study by means of the single charge pulse method showed that the thickness of the passivating oxide films on zirconium, zircaloy-2, titanium and niobium increased during irradiation in dried argon-oxygen mixtures at 20°C. A pure gamma irradiation of these metals in a mixture or argon-oxygen increased the oxide thickness to about the same extent.

2. The influence of surface treatment on the behaviour of carbon steels and stainless steels in water and steam at high temperature.

The influence of surface treatment on the oxidation of metals, as observed in many cases, indicates the usefulness of a systematic study of this factor in the oxidation of steels in high temperature water and steam. This problem has a special importance for the construction of nuclear reactors of the PWR and BWR type.

Two steel varieties have been studied : a low-alloyed carbon steel and a stainless steel type AISI 304. Four surface treatments have been chosen initially : electropolishing in an acetic-perchloric acid bath, under-water mechanical polishing on paper SiC up to grid 600, grinding and milling.

a) In the first part of the study, these steels have been autoclaved in water at 300°C and in steam at 400°C and 200 kg/cm^2 for periods up to 2000 hours.

The results show a very beneficial effect of the electropolishing treatment at both temperatures. The corrosion rate on the electropolished samples is 2 to 3 times lower than on the mechanically prepared samples. The corrosion rate is highest on the milled samples.

As to the stainless steel, the effect of electropolishing is even more marked but only in water at 300°C. In steam at 400°C the mechanically prepared samples turn out to be much better.

Statistical evaluation of a large number of results showed that the observed differences were entirely meaningful.

b) Following these results, a second part of the study comprised the evaluation of the effect of surface treatment on the corrosion of stainless steels for a wider temperature range (200°C to 500°C).

This has been done in order not only to specify the best surface treatment for the use of these steels at different temperatures, but also to gain insight into the oxidation kinetics of 18-8 steels, which change with the mechanical preparation of the surface.

c) At the same time, the different characteristics of the sample surfaces have been defined. The roughness has been measured by means of a profilograph. The thickness of the passivation layers has been measured before introduction of the samples inot the autoclaves, by means of an electrochemical method (coulometry at constant current density). The deformed surface layer has been estimated by following the broadening of diffraction lines.
Finally, the morphology of the oxide layers, formed at different temperatures and after different surface treatment, has been checked by optical and electron microscopy.

These supplementary tests helped in distinguishing the separate influence of each characteristic of the surface on the corrosion.

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Abstract

STUDY OF THE BR2 NUCLEAR CHARACTERISTICS BY MEANS OF ITS MOCK-UP BR02.

F. Motte
Head of the Reactor Physics section, Reactor Study department.

J. Debrue
Senior scientist at the Reactor Physics section, Reactor Study department.

Centre d'Etude de l'Energie Nucléaire (C.E.N.)
MOL (Belgium)

Work performed by the Association C.E.N.-EURATOM for the exploitation of the BR2 reactor and its auxiliary installations. (Contract 006/60/5 BRAB.)

The determination of the nuclear characteristics of the materials testing reactor BR2 is founded essentially on the measurements made in the reactor BR02, the nuclear mock-up of BR2.
The correlation between the two systems, as is described, is based upon a correct forecast of the BR2 behaviour from the tests conducted in BR02.

The increase to full power operation of BR2 has been executed in two stages, one in 1962 and the other in 1963; prior to this, an experimental study has been made in BR02 on the selected core configuration of fourteen fuel elements. This study is presented briefly; it deals

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with the following items : macroscopic flux distributions, fine flux structure, neutron spectra, reactivity value of fuel and poison, reactivity worth of control rods. This information allows to establish by calculation the reactivity balance and the associated control rods shift at any time during the BR2 high flux operation and also contributes to the knowledge of the hot spot factors.

The observations and measurements done in BR2 allow one to verify the established forecasts, and more particularly to evaluate the validity of some simulation models adopted in BRO2. The measurements carried out in BR2 during the high power operation periods and at low power level, during the reactor shut-downs, are described; they lead to the determination of the reactor characteristics evoluation during fuel burn-up : excess of reactivity, flux levels, neutron and gamma heating distributions, neutron spectra and fine flux structure; more particularly and in accordance with one of the more important objectives of the tests, the highest reached operation level, expressed in terms of maximum heat flux, can be deduced.